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THE NRL-EPRI RESEARCH PROGRAM (RP886-2) EVALUATION AND PREDICTI--ETC(U)

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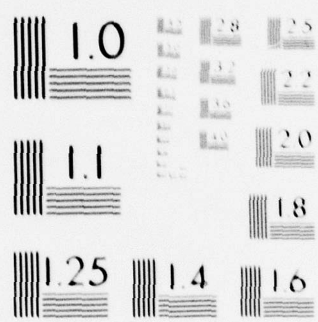

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(10) J. R. Hawthorne, Editor

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Nuclear reactor pressure vessel materials are subject to progressive reductions in fracture resistance in service due to neutron irradiation. Current technology is insufficient to quantitatively predict radiation embrittlement for all typical materials and their metallurgical variations for neutron fluences of interest. Another needed refinement to radiation effects technology involves the establishment of a relationship between apparent notch ductility and fracture toughness in the irradiated condition. The current ~~NRL–EPRI RP886-2~~ program was formulated → next page

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20. Abstract (Continued)

to advance both areas for the benefit of reactor vessel design and operation. An immediate objective involves the development of a high quality data base for the evaluation of current radiation embrittlement projection methods and the development of improved methods.

This report documents program highlights and accomplishments during CY 1977 and plans for the forthcoming second-year investigation.

CONTENTS

THE NRL-EPRI RESEARCH PROGRAM (RP886-2)
EVALUATION AND PREDICTION OF NEUTRON EMBRITTLEMENT
IN REACTOR PRESSURE VESSEL MATERIALS
ANNUAL PROGRESS REPORT FOR CY 1977

SUMMARY

The NRL-EPRI RP886-2 Program was formalized on January 10, 1977. This report documents the highlights and accomplishments of the NRL research efforts and investigations during CY 1977, the first year of program operation.

The initial three-year effort focuses on radiation-induced property changes to reactor pressure vessel materials typical of past as well as current commercial production practices. Radiation effects are being investigated by standard Charpy-V (C_V), fatigue precracked Charpy-V (PCC_V), and compact toughness (CT) test methods. Primary accomplishments during CY 1977 relate to: (1) project design and materials selection, (2) specimen production, (3) acquisition and development of the materials irradiation facility, (4) conduct of the initial materials irradiation experiments, and (5) development of plans for postirradiation materials evaluation.

Research plans and expected accomplishments for CY 1978 are also summarized.

I. INTRODUCTION. J. R. Hawthorne

The Naval Research Laboratory (NRL) and the Electric Power Research Institute (EPRI) have entered into a cooperative research and development program on materials and material applications for nuclear energy systems with emphasis on material safety, reliability, and environment capabilities. The current (initial) effort focuses on radiation-induced property changes to steels and weld metals used in the construction of reactor pressure vessels. Properties under study are notch ductility, fracture toughness, and strength. Primary objectives are: (a) to develop a data base for the evaluation of current radiation embrittlement projection methods and for the development of improved procedures, (b) to investigate the relationship, if one exists, between radiation effects measured by the C_V test method and fracture mechanics test methods, (c) to determine the radiation embrittlement sensitivities of a broad range of reactor pressure vessel materials (plates, forgings, welds), and (d) to assess the effects of selected composition variations.

Note: Manuscript submitted March 7, 1978.

This report summarizes highlights of the first year's effort including project design and materials selection, acquisition and development of the materials irradiation facility, initial irradiation operations, and planning for postirradiation testing and analyses. The present set of investigations has a three-year timeframe with completion scheduled for January 1980.

II. MATERIALS. J. R. Hawthorne

Materials Selection

The selection of materials for the program was accomplished jointly by NRL and EPRI representatives. Eight materials were chosen for the investigations and include two plates of A533-B steel, one plate of A302-B steel, one forging of A508-2 steel, and four submerged arc welds (Table 1). The plates and forgings were among those commercial materials evaluated by the EPRI RP232 Program earlier; accordingly, preirradiation material properties are documented. The A533-B plates, Codes CAB and CBB, permit a comparison of USA and European manufacturing. The A302-B plate, Code N, also is known as the ASTM A302-B reference correlation monitor material (1) which has seen wide usage in reactor vessel surveillance programs. Program objectives for the welds include assessments of the effects of specific composition variations and upper shelf energy variations. That is, the detrimental effects of a high copper content ($\sim$.35% Cu) on radiation resistance compared to an intermediate copper content ($\sim$.18/.20% Cu) are to be investigated (2). Likewise, the significance to radiation resistance of a high (>122J) C_v upper shelf energy (preirradiation) produced by one type of welding flux compared to a relatively low (81 to 95J) C_v upper shelf energy produced by another commonly used flux is to be explored. Three of the experimental welds are being produced by Combustion Engineering, Inc., under EPRI contract, using representative commercial production practices. Arrangements for securing Weld 4 (Table 1) have not yet been completed.

Test Matrix

The irradiation test matrix is illustrated in Table 2 and was developed by NRL and EPRI on 15 April 1977. Each of the twelve irradiation experiments will include standard C_v specimens, fatigue precracked C_v specimens, and 1-in. compact toughness (1TCT) specimens for notch ductility and fracture toughness determinations. The three fluence levels (n/cm^2 >1 MeV) selected for study: 1 to 2 x 10^{18}, $\sim$8 x 10^{18}, and 3 to 4 x 10^{19}, respectively represent initial reactor vessel service, early vessel life, and end-of-life conditions. The 8 x 10^{18} fluence condition was specially chosen for its correspondence with the knee of the radiation embrittlement versus fluence trend curve observed with other comparable materials.

Table 2 indicates a primary program emphasis on Weld 1 and Plate CAB as reference materials. Based on composition, it is anticipated that Weld 1 will exhibit the poorest postirradiation properties and radiation resistance of all the materials for any given fluence condition. Plate

TABLE 1

MATERIALS SELECTED FOR INVESTIGATION

Type	Availability	Status*
A533-B (U.S.,Code CAB)	At NRL	In-Reactor
A533-B (Foreign made, Code CBB)	At NRL	Specs. Machined
A508-2 (U.S., Code BCB)	At NRL	Specs. Machined
A302-B (USS Ref.,Code N)	At NRL	Not cut
S/A Weld #1 (≥.30 Cu, Low Shelf)	Not Availabale	CE Fabricating
S/A Weld #2 (≥.30 Cu, Hi Shelf)	Not Available	CE Fabricating
S/A Weld #3 (.18/.20 Cu, Low Shelf)	Not Available	CE Fabricating
S/A Weld #4 (.18/.20 Cu, Hi Shelf)	Not Available	? CE Fabrication

*As of 31 December 1977.

TABLE 2

IRRADIATION TEST MATRIX AND PRIORITIES

FLUENCE[a] AT 288°C (550°F)

Material	$1\text{-}2 \times 10^{18}$	$\sim 8 \times 10^{18}$	$3\text{-}4 \times 10^{19}$
A533-B (CAB)		X (#1)[b]	X (#2)
A533-B (CBB)			X (#3)
A508-2 (BCB)			X (#4 or 5)
A302-B (USS, N)		X (#4 or 5)	
Weld 1 (≥.30 Cu, Low Shelf)	X	X	X
Weld 2 (≥.30 Cu, Hi Shelf)	(a)[c]	X	
Weld 3 (.18/.20 Cu, Low Shelf)		X (#4 or 6)	(a)
Weld 4 (.18/.20 Cu, Hi Shelf)			X

[a] n/cm^2 >1 MeV.

[b] Irradiation Experiment Priority.

[c] An option considered for Experiment No. 12.

CAB, on the other hand, represents improved production material and should exhibit relatively high radiation resistance. As part of the overall materials plan, thermal control tests will be conducted to reveal the effects of 288°C temperature conditioning in the absence of irradiation. The plan is to limit such evaluations to that aging condition corresponding to the longest of the reactor irradiation exposures.

Material Availability and Specimen Cutting

The three plate materials and the forging material only have been received at NRL thus far (Table 1). With the exception of Plate Code N, specimen cutting and machining operations have been completed for the available material. In the case of thermal control specimens, final machining will be accomplished after temperature conditioning. In addition, fatigue precracking operations on PCC_v and CT specimens have been undertaken on an "as-needed" basis to retain program flexibility.

The delivery of Weld 1 will be required by 1 March 1978 to maintain the planned materials irradiation schedule (see Section III below). Later deliveries of Welds 2, 3, and 4 are permitted. However, it should be recognized that the conduct of preirradiation condition testing in advance of irradiation experiments is highly desirable, especially in the case of newly fabricated material such as the welds.

III. REACTOR FACILITY AND OPERATIONS. H. E. Watson and J. R. Hawthorne

Facility Acquisition

The schedule of events leading up to the acquisition of the irradiation facility for the EPRI Program is detailed in Fig. 1. Although the NRL-EPRI contractual agreement was not finalized until January 1977, the initial process of securing and qualifying a new facility began in May 1976.

The first site considered for the NRL-EPRI irradiation program was the Massachusetts Institute of Technology (MIT) Reactor. This reactor was ultimately rejected because of (a) excessive gamma heating, (b) irradiation space limitations, (c) poor access to the core (sealed core), (d) high cost, and (e) an undesirable location for the control console (i.e., contaminated area).

Upon rejection of the MIT Reactor, other options were investigated. The several reactor facilities considered included those at the University of Virginia, the University of North Carolina, Georgia Institute of Technology, the University of Missouri,Lowell Technical Institute, the Industrial Reactors Laboratory, the Union Carbide Research Center, and the Oak Ridge National Laboratory (ORNL). When all factors were evaluated, the Bulk Shielding Reactor (BSR) at ORNL appeared to be the best reactor for the irradiation program.

The BSR is a government-owned facility; therefore, it cannot compete

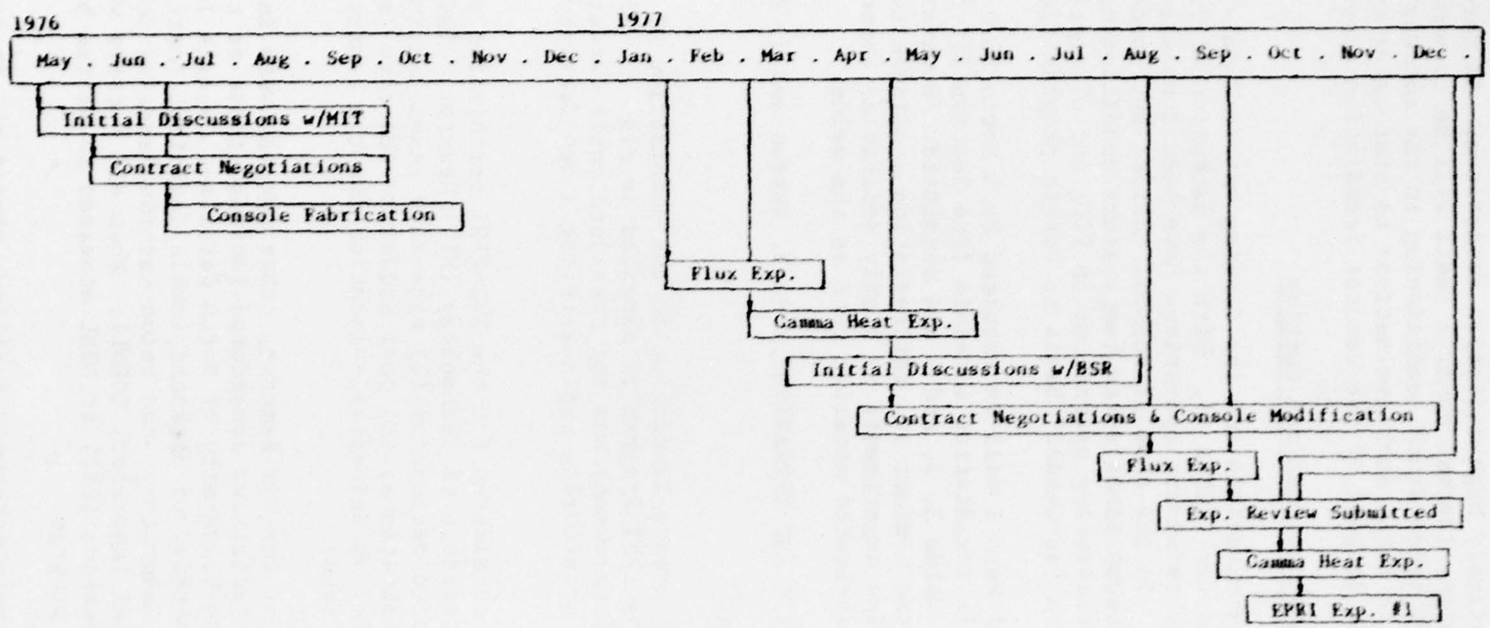


Fig. 1 - Schedule of events

with privately-owned reactor facilities. Before a contract could be established for irradiation services, it was necessary to prove to the Department of Energy that private facilities which could meet the requirements of this program were not available. This was accomplished and a contract was established in May 1977.

Upon finalization of the contract, an experiment review questionnaire was submitted to ORNL Reactor Safeguards Committee. This Committee has the responsibility for all experiment designs and temperature control concepts as well as control consoles and must give its approval before any irradiation experiment can be performed. A personal appearance before the Board was necessary before final approval was granted the NRL experiment design and external control system. Prior to this meeting, the proposed NRL console was modified to meet the known requirements of ORNL and shipped to the reactor in preparation for installation.

Official approval of the Committee was secured on 20 December 1977; the preliminary gamma heat survey experiment was inserted in the reactor on the same day. A special neutron flux mapping experiment, described in the next section, was completed in advance of this survey. On 28 December the gamma heat survey experiment was discharged and immediately replaced with the first NRL-EPRI materials irradiation experiment. This initial experiment has confirmed the adequacy of the irradiation assembly and control system design.

Neutron Flux Survey

In accordance with NRL routine procedures for qualifying a new irradiation facility, a neutron flux survey of the proposed BSR facility (BSR 67) was performed. The objectives were threefold: (a) to establish the ambient neutron flux levels in the facility, (b) to identify the vertical and horizontal flux gradients, and (c) to determine the proper elevation of the specimen assembly in the reactor core piece to receive the most uniform neutron exposure. Normally, information on neutron flux conditions in individual facilities, available from reactor operations, represents either flux approximations or extrapolations from nearly facilities and cannot be relied upon fully.

The experimental unit designed for the flux survey consisted of two blocks of carbon steel simulating the active specimen volume in the materials irradiation assembly and three aluminum blocks (Figure 2). The aluminum blocks served to displace the reactor coolant water from that volume normally taken up by voids in the irradiation assembly. The longitudinal holes at corner and center positions received the flux monitor tubes containing Fe, Ni, CoAl, and AgAl wires.

The initial survey irradiation involved a 61,350 kilowatt hour exposure. The dosimeter wires were analyzed at the Radiation Counting Laboratory of EG&G Idaho, Inc. In Fig. 2, the dosimetry results clearly indicate that the materials irradiation assembly should be positioned two inches higher than the initial survey position. At this new location,

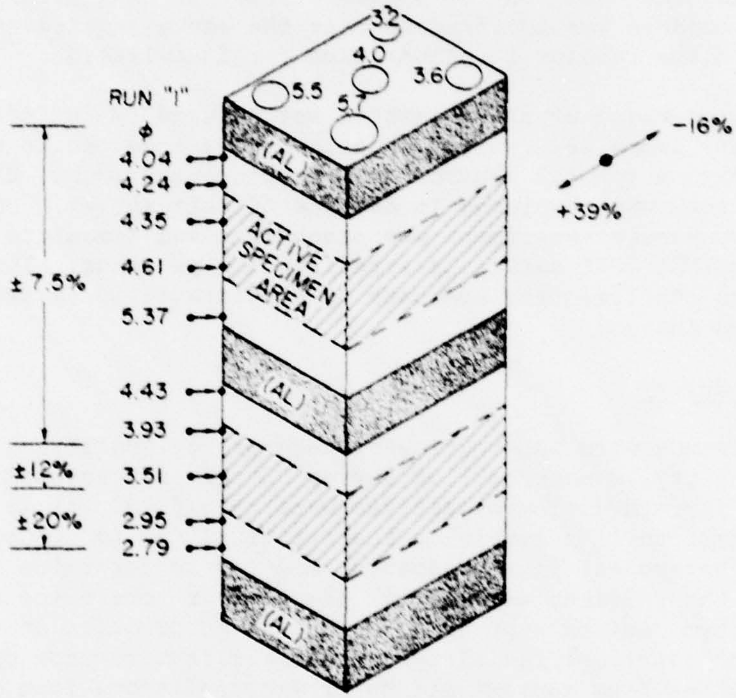


Fig. 2 - Schematic design of experimental unit developed for neutron flux surveys in the BSR. Flux monitor tubes were placed in the longitudinal holes located at the assembly corners and center. Flux intensities (n/cm^2-sec >1 MeV x 10^{12}) over the assembly length in the center monitoring location are also shown.

the neutron flux over the active specimen volumes will be between 4.0 and 4.5 x 10^{12} n/cm^2-sec (>1 MeV) according to the center monitor results. Also, the vertical flux gradient over most of the assembly should be within 10 percent of the average.

Because of a misunderstanding, the specified fuel configuration about the experimental facility was not provided for the flux survey experiment. That is, fuel was provided on only three and not all sides of the facility as required. As a result, a rather large horizontal flux gradient is noted from the dosimetry data. A repeat flux survey irradiation, using the requested fuel configuration, has since been conducted; however, the results are not yet available.

In addition to selected fuel adjustments, an alternate technique for balancing neutron fluences is often used by NRL. This method involves the rotation of the experimental assembly by 180 degrees at a point midway through the scheduled irradiation period. Because of the BSR reactor core piece configuration and space limitations however, only the upper half of the NRL-EPRI experiments can be rotated for this purpose. Nevertheless, the multi-specimen nature of the assemblies makes it possible to minimize gradient effects by experimental arrangement.

Gamma Heat Survey

Paralleling the neutron flux survey, a gamma heat survey is required for a new irradiation facility to complete essential information on ambient exposure conditions. Objectives of the survey in this case are (a) to qualify the gamma heating conditions which must be handled by the experiment temperature control system, and (b) to assess the heating gradients which must be overcome by the irradiation assembly design. The NRL approach entails the construction and irradiation of a highly instrumented specimen assembly of the proposed design evolved from gamma heating projections. In this instance, the proposed specimen layout for upper and lower experiment units is shown schematically in Fig. 3. The upper unit contains 4 1TCT, 1 tensile, and 6 C_V specimens; the lower unit contains 3 1TCT, 1 tensile, 6 C_V and 10 PCC_V specimens. To facilitate heat transfer between specimens, each specimen array is unitized through use of a common steel framework and clamping arrangement.

The first attempt to survey the gamma heating in the BSR facility was only partially successful in that one of the two experiment units experienced a leak in the outer containment. Although the inner containment remained intact, the leak precluded any controlled temperature operation of that unit. (The leak was subsequently traced to a longitudinal seam weld failure introduced by a last minute, on-site addition of welded spacer pieces. This was done at the request of the BSR Safety Review Board). The performance of the lower unit, however, did prove that the temperature control system was capable of handling the proposed irradiation experiment needs under reactor startup conditions and under normal operating conditions. The lower unit also indicated that the experiment design was generally satisfactory with regard to specimen

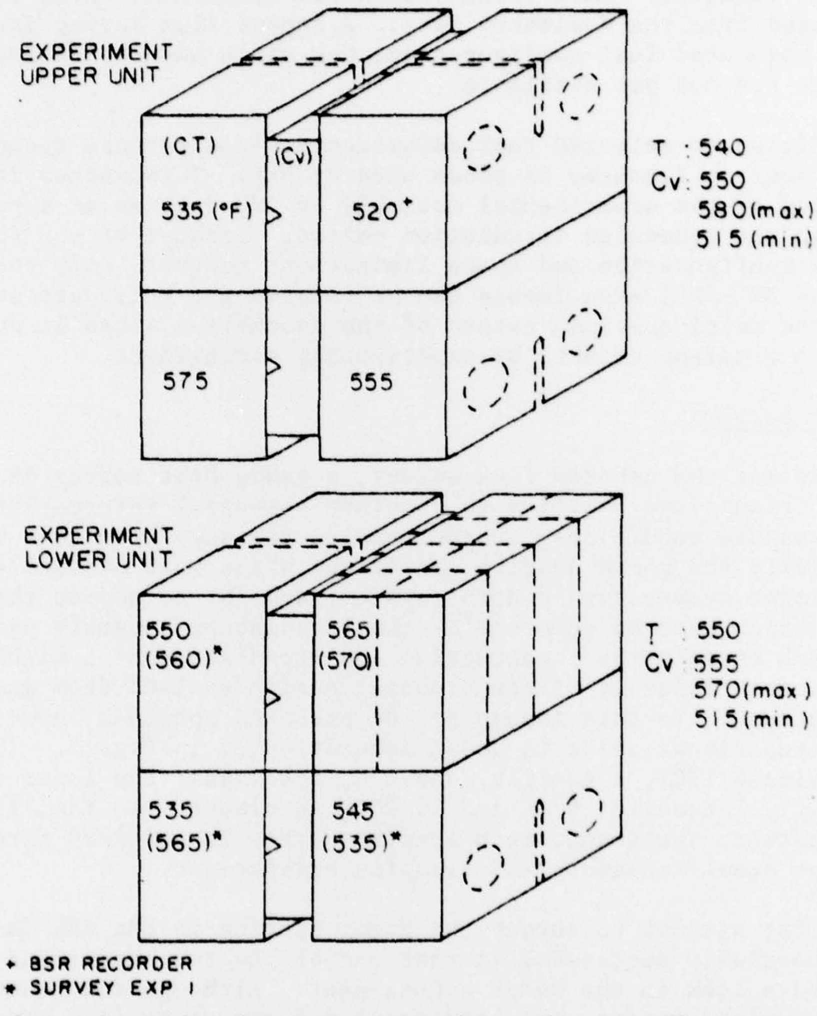


Fig. 3 - Schematic showing proposed specimen configuration in the material irradiation experiment. Temperature patterns observed with the lower unit of the gamma heat survey experiment (numbers in parentheses) and with the initial materials irradiation experiment are shown. The one temperature determination marked (BSR recorder) is in question because of uncertain instrument calibration.

temperature patterns. Based on these tentative observations, a decision was made to proceed with the same design for the initial material irradiation experiment. System temperature control capabilities and assembly temperature patterns were subsequently confirmed. Figure 3 compares temperatures obtained with survey experiment No. 1 (numbers in parentheses) and materials irradiation experiment No. 1. Based on the latter, the irradiation assembly design is now being fine-tuned for optimum temperature uniformity.

Irradiation Schedule

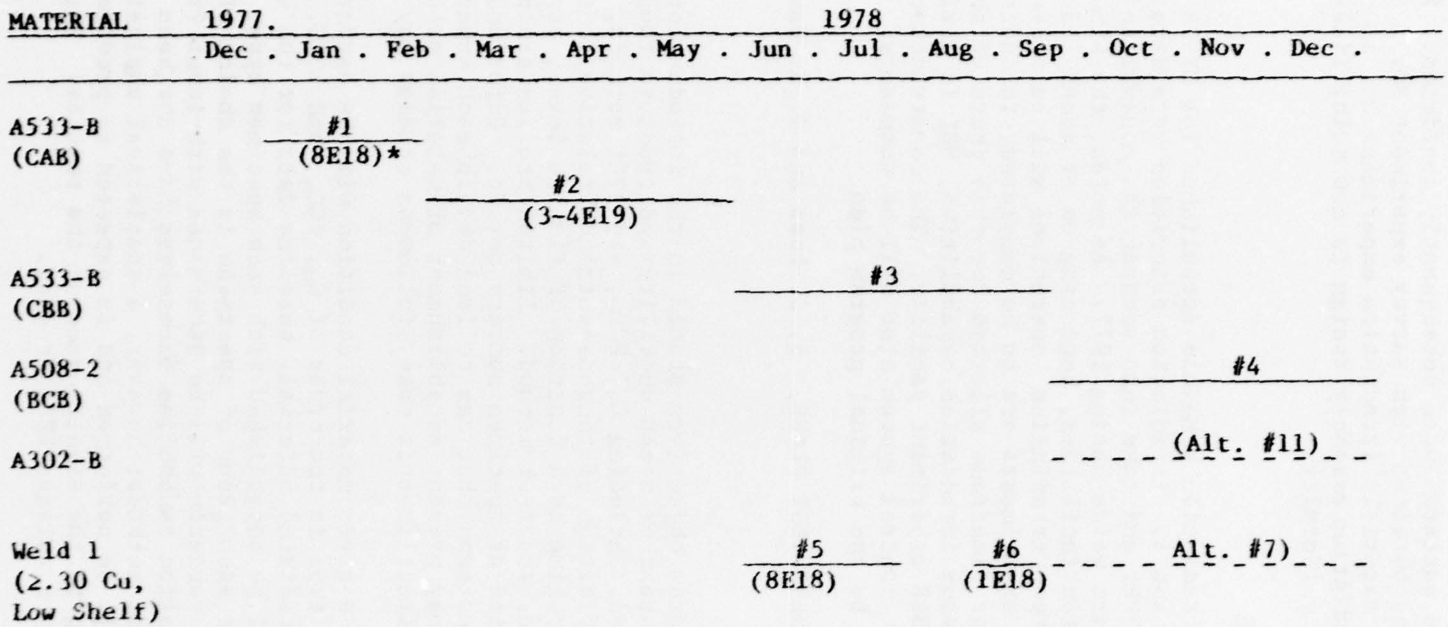
The materials irradiation schedule established for CY 78 and CY 79 is shown in Figs. 4 and 5. Irradiation priorities were determined jointly by NRL and EPRI and take into account the nonavailability of the four submerged arc welds during 1977. As noted, the simultaneous use of two irradiation facilities, beginning on or about 1 June 1978, is planned. Concurrent irradiation operations will be necessary if all twelve required experiments are to be completed, including testing, within the three-year timeframe allotted for this phase of the study. To provide simultaneous irradiation capabilities, NRL is making arrangements for a second BSR experiment facility. The construction of a duplicate experiment control system also will be necessary. Twin systems were not called for by the original program plan.

IV. FRACTURE TOUGHNESS TEST PLANS. F. J. Loss and J. R. Hawthorne

Overview

The primary program objectives stated in the introduction are amplified here in the context of notch ductility and fracture toughness test methods to be applied, including C_V, PCC_V, and 1TCT methods. The major emphasis is upon definition of toughness trends relative to the brittle-ductile transition regime as a function of fluence level, product form, chemical composition, and test method. Limited studies are planned for the upper shelf regime as specimen numbers permit. Unfortunately, the small numbers of specimens that can be included in each experimental irradiation assembly may prevent establishment of material property trends to the accuracy desired; in this case, follow-on studies may be required.

The testing for a given material condition will be performed with respect to specimen type in the order of C_V, PCC_V, and 1TCT. Before investigating an irradiated material, baseline data for the unirradiated condition first will be established with each specimen type. The independent variable for each group of specimens is the choice of test temperatures. This variable will be determined with input from a statistician. The irradiation assemblies themselves have not been designed purely by statistical methods; however, a statistical analysis of all the results can still be performed and is expected to provide additional information relating to the significance of the results. This analysis will be performed by a related EPRI program.



*Fluence (8×10^{18} n/cm^2 >1 MeV).

Fig. 4 - Experiment irradiation schedule for CY 1978

MATERIAL	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
Weld 1 (≥.30 Cu, Low Shelf)		#7 (3-4E19)*										
Weld 2 (≥.30 Cu, Hi Shelf)					#8							
Weld 3 (.18/.20 Cu, Low Shelf)					#9							
Weld 4 (.18/.20 Cu, Hi Shelf)							#10					
A302-B							#11					
MATERIAL CONDITION "XYZ"			? #12									

*Fluence (3 to 4 x 10^{19} n/cm^2 >1 MeV)

Fig. 5 - Experiment irradiation schedule for CY 1979

Charpy-V Tests

The objective of the C_v tests is to assess the embrittlement characteristics of each material condition by a conventional method and thereby provide an index of the data from this program to procedures (3,4) currently used in fracture safety analyses. These tests, when augmented by drop weight tests of the unirradiated condition,* will define the radiation-induced elevation in the reference temperature, RT_{NDT}. as used in ASME Section III. Assessments of the degradation in upper shelf energy with irradiation will also be performed. As a major objective, these trends will provide a data base with which to evaluate embrittlement projections set forth by NRC Regulatory Guide 1.99 (4).

There will be twelve C_v specimens available from each irradiation experiment. The test temperatures for these specimens will be chosen so as to define the full C_v curve that includes both the transition and upper shelf regimes. Tentatively, three C_v specimens only will be allocated for upper shelf regime tests. Depending on material preirradiation upper shelf level (high or low), the first four C_v specimens of each experiment will be tested as follows, based on the estimated postirradiation transition behavior:

Material Type	Material Codes	Test Temp. Equivalent
High Upper Shelf	CAB, CBB, BCB, W2, W4	C_v 27-41 J (20-30 ft-lb)
		C_v 68-81 J (50-60 ft-lb)
		C_v 102-122 J (75-90 ft-lb)
		C_v upper shelf
Low Upper Shelf	N, W1, W3	C_v 14-27 J (10-20 ft-lb)
		C_v 41-48 J (30-35 ft-lb)
		C_v 61-75 J (45-55 ft-lb)
		C_v upper shelf

The remaining specimens will be applied to delineate the full C_v curve with particular attention given to establishing the C_v 41 J (30 ft-lb) and C_v 68 J (50 ft-lb) transition temperatures. Unless data scatter is small, upper shelf tests will be conducted at one temperature only. The estimation of postirradiation behavior in advance of testing will be guided by prior NRL experience on similar materials and material compositions, and by currently-used methods for projecting radiation embrittlement (4,5). Schedule permitting, specimen testing will not be initiated until neutron dosimetry results for that experiment are available.

The hot cell machine to be used for all postirradiation C_v and PCC_v tests was rechecked for calibration in August 1977 using specimens supplied by the Army Materials and Mechanics Research Center (AMMRC).

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* The drop weight test identifies the material nil ductility transition (NDT) temperature.

Based on the results, AMMRC has certified the machine as acceptable for inspection testing for a period of one year (6). Similar calibration tests are scheduled for the relocated out-of-cell machine for pre-irradiation (reference condition) C_V and PCC_V tests. Rechecks of calibration will be performed in the future as necessary.

Precracked Charpy-V Tests

These specimens will be tested dynamically in accordance with procedures developed in an earlier EPRI program (7). Each irradiation experiment will provide ten PCC_V specimens. A plan for the specific test temperatures has not yet been formalized;* however, tests will focus on the transition regime as in the C_V test plan above. It is anticipated that the PCC_V specimen can define dynamic fracture toughness (K_{Id}) in the transition region to a value of approximately 110 $MPa\sqrt{m}$ (100 $ksi\sqrt{in.}$), using the J-integral method up to the point of maximum load. The specimen size is too small to measure a valid fracture toughness (by ASTM E-399 criteria for static tests) above a value of approximately 44 $MPa\sqrt{m}$ (40 $ksi\sqrt{in.}$). Consequently, the J-integral approach must be used to project a K_{Jd} from these tests for levels in excess of this value. This procedure is believed to be acceptable only when fracture initiates in a cleavage mode. This type of fracture can be easily interpreted from available instrumentation and will define the point of crack initiation (i.e., a single specimen J_{Id} test).

The K_{Id} curve thus established will be assessed in terms of the K_{IR} curve of ASME Section III. In addition, the results will enable a comparison to be made between the embrittlement trend projected by the C_V test and that defined by a fracture mechanics-type test that is more directly relatable to the critical flaw size and stress level in a structure. Unfortunately, the limited thickness of the PCC_V specimen will permit definition of only the beginning of the postirradiation K_{Id} curve in the transition region. Nevertheless, the measured trends will permit a partial verification of the temperature shift of the K_{IR} curve with irradiation that is currently determined from the C_V specimen in terms of ΔRT_{NDT}.

Compact Toughness Tests

These specimens will be tested in the static mode with the primary objective being to define the postirradiation K_{Ic} curve in the tranition region. Besides defining this behavior for irradiated material, the results will permit conclusions to be drawn as to whether or not the temperature shift (ΔT) between unirradiated and irradiated condition C_v, or PCC_v, curves (dynamic tests) is identical to the temperature shift between the irradiated and unirradiated condition K_{Ic}

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* The plan for postirradiation PCC_V testing currently is being evolved at NRL.

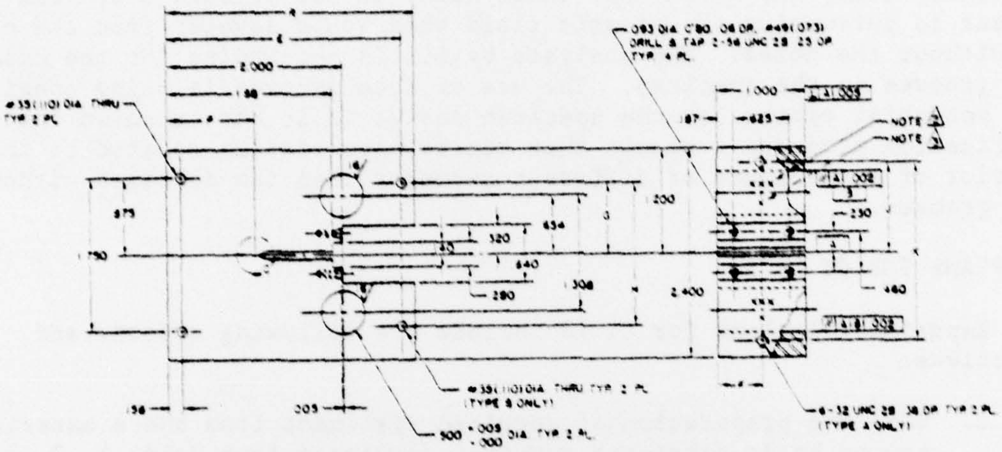
curves (static test). Current application of NRC Regulatory Guide 1.99 assumes the latter correlation to be 1:1. However, the data in Guide 1.99 relate strictly to the irradiation-induced shift in the K_{IR} curve, which is generally taken to be based upon dynamic toughness data.

Each irradiation experiment includes seven 1TCT specimens (Fig. 2). The current plan for material of high upper shelf energy is to test five specimens at four temperatures in the transition region and hold the remaining two in reserve. A decision to commit these specimens to the transition or upper shelf regions will be made upon review of the results. For the materials of low upper shelf energy, it is planned to test four specimens at four temperatures in the transition region, test one specimen on the upper shelf, and hold the remaining specimens in reserve.

Test temperatures in the transition region will be chosen initially to result in K_{Ic} values of 55, 110, 165, and 220 $MPa\sqrt{m}$ (50, 100, 150, and 200 $ksi\sqrt{in.}$) in the transition region, thereby defining the toughness range of major interest. The K_{Ic} curve for the unirradiated material will provide guidance as to the slope of the postirradiation K_{Ic} curve whereas the temperature shift between the unirradiated and irradiated condition C_v curves will provide an estimate for the absolute position, temperature wise, of the postirradiation K_{Ic} curve.

It is currently proposed to test only the low shelf material on the upper shelf. From past experience it is judged that the high shelf materials will exhibit such a high upper shelf toughness as to be more than sufficient to meet the requirements of an accident analysis. Therefore, it appears more productive to commit the limited specimen inventory to a better definition of toughness trends in the transition region. For tests that are conducted on the upper shelf, a primary objective will be to express the toughness in terms of C_v energy. This information, when augmented by additional data from future tests, will enable a quantitative interpretation of the upper shelf energy from C_v specimen results of reactor surveillance capsules.

The 1TCT specimens will be investigated in terms of the J-integral approach using the unloading-compliance method. In this way, K_{Ic} values can be interpreted from the J_{Ic} values when the E-399 thickness criterion for valid K_{Ic} tests has been violated. In some cases a K_{Ic} test may result directly from tests at low temperatures in the transition region. In other cases the fractures may exhibit a cleavage initiation while violating the E-399 linearity requirements of the load versus deflection record. When this occurs the specimen becomes a single specimen J_{Ic} test, as previously discussed for the PCC_v tests, and a K_{Jc} value can be predicted. At higher temperatures in the transition region and on the upper shelf, crack initiation will be by ductile tearing so that a J-R curve approach must be used to define J_{Ic}. Due to the limited specimen numbers available, it is essential that this R curve be established from a single specimen test which is possible with the unloading-compliance technique.



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2. CODE NO.

3. CONSECUTIVE NO.

Fig. 6 - Compact Toughness (CT) Test Specimen Design

In order to apply the unloading-compliance method, the notch region of standard E-399 specimen design has been modified to permit the use of knife blades to minimize friction in the measurement of load-line deflection. The specimen design is illustrated in Fig. 6. In some cases, the specimen thickness will be reduced from 25.4 to 24.1 mm (1.00 to 0.95 in.) to permit a greater flexibility in the BSR irradiation assembly design. This small change is not expected to have an appreciable effect on the results. An additional specimen modification necessitated by the irradiation assembly design was the inclusion of two small holes, parallel to the holes for the specimen loading pins, but above and below the path of crack propagation (see Fig. 6). An analysis by the Lawrence Livermore Laboratory (LLL) has shown that these holes do not present a problem with respect to perturbing the plastic field that would develop from the notch tip without the holes. The analysis by LLL is continuing for the case of face grooves in the specimen. The use of face grooves is being considered as a potential option for the specimen design if it can be shown that this modification produces a result that can be more readily related to the behavior of a structure of different geometry than the specimen without face grooves.

V. PLANS FOR CY 1978

Experimental plans for CY 78 include the following efforts and objectives:

1. Complete preparation of required specimens from those materials now on hand; fabricate required specimens from Welds 1, 2, 3, and 4 as they become available.

2. Acquire and develop a second BSR experiment facility to permit simultaneous irradiation operations.

3. Construct and irradiate materials experiments numbers 1 through 7 in accordance with established priorities.

4. Commence postirradiation testing of materials experiments numbers 1 to 3, 5 and 6.

5. Conduct preirradiation condition (reference) testing, as required, on materials now available.

Research findings and accomplishments during this period will be documented in the second annual progress report.

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